**Tennessee Valley Authority**, Post Office Box 2000, Soddy-Daisy, Tennessee 37379-2000

December 14, 2011

10 CFR 50.73

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Sequoyah Nuclear Plant, Unit 1
Facility Operating License No. DPR-77
NRC Docket Nos. 50-327

Subject: **License Event Report 327/2011-005, "Reactor Trip as a Result of Reactor Coolant Pump Bus Undervoltage," Revision 1**

Reference: Letter from TVA to NRC, "License Event Report 327/2011-005, "Reactor Trip as a Result of Reactor Coolant Pump Bus Undervoltage", dated October 14, 2011

The purpose of this letter is to provide the enclosed revision to the referenced licensee event report (LER). The referenced LER has been revised with supplemental information concerning an automatic reactor trip and automatic engineered safety feature actuation of auxiliary feedwater following an undervoltage condition of two reactor coolant pumps. Changes to the previous report are indicated by revision bars on the right side margin of the page.

The Tennessee Valley Authority is submitting this report in accordance with 10 CFR 50.73(a)(2)(iv)(A), a condition that resulted in automatic actuation of the reactor protection system and the auxiliary feedwater system.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact G. M. Cook, Sequoyah Site Licensing Manager, at (423) 843-7170.

Respectfully,

John T. Carlin
Site Vice President
Sequoyah Nuclear Plant

cc: NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

IE22
NRR

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/20

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resources@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

# LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. PAGE |
|---|---|---|
| Sequoyah Nuclear Plant Unit 1 | 05000327 | 1 OF 6 |

**4. TITLE:**
Reactor Trip as a Result of Reactor Coolant Pump Bus Undervoltage - Revision 1

| 5. EVENT DATE | | | 6. LER NUMBER | | | 7. REPORT DATE | | | 8. OTHER FACILITIES INVOLVED | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REV NO. | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER |
| 08 | 18 | 2011 | 2011 | - 005 - | 01 | 12 | 14 | 2011 | FACILITY NAME | DOCKET NUMBER |

**9. OPERATING MODE**
1

**10. POWER LEVEL**
100

**11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §:** *(Check all that apply)*

- ☐ 20.2201(b)
- ☐ 20.2201(d)
- ☐ 20.2203(a)(1)
- ☐ 20.2203(a)(2)(i)
- ☐ 20.2203(a)(2)(ii)
- ☐ 20.2203(a)(2)(iii)
- ☐ 20.2203(a)(2)(iv)
- ☐ 20.2203(a)(2)(v)
- ☐ 20.2203(a)(2)(vi)
- ☐ 20.2203(a)(3)(i)
- ☐ 20.2203(a)(3)(ii)
- ☐ 20.2203(a)(4)
- ☐ 50.36(c)(1)(i)(A)
- ☐ 50.36(c)(1)(ii)(A)
- ☐ 50.36(c)(2)
- ☐ 50.46(a)(3)(ii)
- ☐ 50.73(a)(2)(i)(A)
- ☐ 50.73(a)(2)(i)(B)
- ☐ 50.73(a)(2)(i)(C)
- ☐ 50.73(a)(2)(ii)(A)
- ☐ 50.73(a)(2)(ii)(B)
- ☐ 50.73(a)(2)(iii)
- ☒ 50.73(a)(2)(iv)(A)
- ☐ 50.73(a)(2)(v)(A)
- ☐ 50.73(a)(2)(v)(B)
- ☐ 50.73(a)(2)(v)(C)
- ☐ 50.73(a)(2)(v)(D)
- ☐ 50.73(a)(2)(vii)
- ☐ 50.73(a)(2)(viii)(A)
- ☐ 50.73(a)(2)(viii)(B)
- ☐ 50.73(a)(2)(ix)(A)
- ☐ 50.73(a)(2)(x)
- ☐ 73.71(a)(4)
- ☐ 73.71(a)(5)
- ☐ OTHER
  Specify in Abstract below or in NRC Form 366A

**12. LICENSEE CONTACT FOR THIS LER**

| FACILITY NAME | TELEPHONE NUMBER *(Include Area Code)* |
|---|---|
| SQN - Norm Thomas | (423) 843-7749 |

**13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT**

| CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| E | EC | FU | B569 | Y | | | | | |

**14. SUPPLEMENTAL REPORT EXPECTED**

☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO

**15. EXPECTED SUBMISSION DATE**

| MONTH | DAY | YEAR |
|---|---|---|
| | | |


ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On August 18, 2011, at approximately 2250 Daylight Saving Time (DST), Sequoyah Nuclear Plant (SQN) Unit 1 automatically tripped as a result of reactor coolant pump (RCP) 1 and 3 bus undervoltage. The undervoltage condition on the two unit boards that feed RCPs 1 and 3 was caused by a slow transfer of the 1A Start Bus from normal to alternate supply. A failure of the 1A Start Bus potential transformer (PT) secondary side fuse initiated the slow transfer of the 1A Start Bus.

Unit 1 was stabilized in hot standby following the automatic reactor trip. The cause of the fuse failure was determined to be mechanical separation of end plate from end cap due to inadequate manufacturing aggravated by age. The root cause was determined to be no preventative maintenance to address periodic replacement of fuses for the Start Bus protective relay circuits. The corrective actions will include creating preventative maintenance procedures to address replacing fuses in locations that are trip risk sensitive or could result in engineered safety feature actuations.

| 1. FACILITY NAME | 2. DOCKET | 6. LER NUMBER | | | 3. PAGE |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REV NO. | |
| Sequoyah Nuclear Plant Unit 1 | 05000327 | 2011 | -- 005 | -- 01 | 2 OF 6 |

NARRATIVE

I. PLANT CONDITION(S)

At the time of the event, Sequoyah Nuclear Plant (SQN) Unit 1 was operating at approximately 100 percent rated thermal power.

II. DESCRIPTION OF EVENT

A. Event:

On August 18, 2011, at approximately 2250 Daylight Saving Time (DST), Sequoyah Nuclear Plant (SQN) Unit 1 automatically tripped as a result of reactor coolant pump (RCP) [EIIS Code AB] 1 and 3 undervoltage. The RCP 1 and 3 undervoltage condition was caused by a slow transfer of the 1A Start Bus [EIIS Code EA], which feeds the unit boards that supply the RCPs. The transfer of the 1A Start Bus was caused by a failed fuse in the associated potential transformer (PT) circuit.

The 1A Start Bus PT secondary fuse on C phase opened without the presence of an actual overcurrent condition. The open fuse caused the undervoltage relays 27B-1 and 27B-3 to actuate. Relay 27B-1 opened the 1A Start Bus normal feeder breaker at approximately 90% of normal voltage. Relay 27B-3 provided annunciation of undervoltage to the main control room (MCR). Relay 27R-B closed the 1A Start Bus alternate feeder breaker at approximately 30% of normal voltage. The degraded voltage associated with this transfer resulted in an undervoltage condition on RCPs 1 and 3. This condition resulted in a RCP Undervoltage Reactor Trip signal.

The Start Bus PT fuse that failed was a Bussman Limitron KTN-6. Investigation determined that the fuse failed due to mechanical separation of end plate from end cap due to inadequate manufacturing aggravated by age.

Following the reactor trip, operations noticed that the Turbine Stop Valve #2 [EIIS Code TA] bi-stable light was not lit on the main control room panel. Investigation determined that the turbine valve had closed, but the valve actuator arm on the limit switch was loose, and the limit switch had not actuated. Since the limit switch had not actuated, no signal was sent to the Solid State Protection System (SSPS) [EIIS Code JC] that the turbine valve had closed. This condition would have prevented a Turbine Stop Valve Closure Turbine Trip signal from initiating a reactor trip signal above permissive P-9, which is equivalent to 50 percent reactor power. This trip is not assumed in any Updated Final Safety Analysis (UFSAR) Chapter 15 accident analysis. As such, this condition is not considered a safety system function failure. In addition, the redundant Low Fluid Oil Pressure Turbine Trip signal was verified to operate correctly during this event. The Turbine Stop Valve #2 limit switch was repaired and tested successfully before the unit was restarted.

Following the reactor trip, the auxiliary feedwater system [EIIS Code BA] automatically actuated as expected on loss of the main feedwater pumps [EIIS Code SJ]. The main feedwater pumps were available for recovery using approved plant procedures following the reactor trip. The auxiliary feedwater and steam dump [EIIS Code SG] systems operated as expected to remove decay heat and stabilize the reactor coolant system at the no-load value of 547 degrees Fahrenheit.

The Tennessee Valley Authority is submitting this report in accordance with 10 CFR 50.73(a)(2)(iv)(A), a condition that resulted in automatic actuation of the reactor protection system and the auxiliary feedwater system.

B. Inoperable Structures, Components, or Systems that Contributed to the Event:

None.

C. Dates and Approximate Times of Major Occurrences:

| Date | Description |
|---|---|
| August 18, 2011 at 2250 DST | A failure of the 1A Start Bus potential transformer fuse initiated a slow transfer of the 1A Start Bus from normal to alternate supply. Unit 1 automatically tripped as a result of RCP 1 and 3 undervoltage. |
| August 18, 2011 at 2251 DST | Unit 1 enters mode 3. Operations entered Emergency Procedure E-0 "Reactor Trip or Safety Injection." |
| August 18, 2011 at 2255 DST | Operations entered Emergency Subprocedure ES-0.1, "Reactor Trip Response." |
| August 18, 2011 at 2305 DST | Operations entered Abnormal Operating Procedure (AOP) - I.07, "Turbine Auto Stop Oil Pressure Instrument Malfunction," because the Turbine Stop Valve #2 bi-stable light was not lit. |
| August 19, 2011 at 0114 DST | Operations exits AOP - I.07, "Turbine Auto Stop Oil Pressure Instrument Malfunction," following removing the Turbine Stop Valve #2 limit switch from service. A work order is initiated for the bi-stable light not being lit. |

D. Other Systems or Secondary Functions Affected:

Following the reactor trip, Operations noted that the 1A Condenser Circulating Water (CCW) Pump [EIIS Code KE] had tripped. A work order was initiated to

investigate. Adequate CCW flow was maintained for condenser vacuum by the 1B and 1C CCW pump operation.

E. Method of Discovery:

Control room alarms alerted operators to the start of the event.

F. Operator Actions:

Operations responded to the reactor trip by performing actions in accordance with Emergency Procedure E-0, "Reactor Trip or Safety Injection," and Emergency Subprocedure ES-0.1, "Reactor Trip Response." ES-0.1 is a subprocedure of procedure E-0. Operations performed AOP- I.07, "Turbine Auto Stop Oil Pressure Instrument Malfunction," due to the Turbine Stop Valve #2 bi-stable light not being lit. The operations crew responded to the event as expected.

G. Safety System Responses:

With the exception of the Turbine Stop Valve # 2 limit switch which did not actuate, the plant responded as expected for the conditions of the reactor trip.

III. CAUSE OF THE EVENT

A. Immediate Cause:

The immediate cause of the reactor trip was a failure of the Start Bus 1A potential transformer secondary side fuse because of mechanical separation of end plate from end cap due to inadequate manufacturing aggravated by age.

B. Root Cause:

The root cause was determined to be no preventative maintenance to address periodic replacement of fuses for the Start Bus protective relay circuits.

IV. ANALYSIS OF THE EVENT

Prior to the event, SQN Unit 1 was operating in Mode 1 at approximately 100 percent power and reactor coolant system (RCS) pressure was at the normal operating pressure of 2235 pounds per square inch gauge (psig). Following the reactor trip, while in Mode 1, pressurizer pressure was maintained above the Technical Specification (TS) 3.2.5, "DNB Parameters," limit of equal to or greater than 2220 pounds per square inch atmospheric. Following the reactor trip, RCS pressure rapidly decreased due to the decreasing RCS average temperature and the associated shrinking of coolant volume. The minimum RCS pressure was approximately 2030 psig, well above the pressure that

would have initiated a safety injection signal (1870 psig). Pressurizer pressure recovered gradually, rising to 2264 psig before dropping back to normal operating pressure.

RCS average temperature was at 577 degrees Fahrenheit (F) prior to the reactor trip. After the reactor tripped and during the short excursion from Mode 1, the TS 3.2.5 limit for RCS average temperature of less than or equal to 583 degrees F was not exceeded. The loss of nuclear heat generation resulted in a decrease in RCS temperature to 539 degrees F. The auxiliary feedwater and steam dump systems operated as expected to remove decay heat and stabilize the RCS at the no-load value of 547 degrees F. The main feedwater pumps were available for recovery using approved plant procedures following the reactor trip.

The indicated flow on Reactor Coolant Pumps 1 and 3 decreased by approximately six percent for about two seconds around the time of the reactor trip. However, forced flow was maintained by all four RCPs as indicated by the loop flow transmitters and also by a lack of change in loop temperatures. The limiting TS 3.2.5 RCS total flow rate prior to and after the reactor trip was maintained above the limit of 360,100 gallons per minute. The UFSAR event most similar to this reactor trip is the Partial Loss of Forced Reactor Coolant Flow event described in UFSAR Section 15.2.5. In the analysis of this event, a partial loss of flow involving loss of two reactor coolant pumps was assumed. The resultant RCS flow rate prior to the reactor trip was greater than RCS flow rate assumed in the Partial Loss of Forced Reactor Coolant Flow analysis.

The plant responded as expected for the conditions of the trip. No TS limits were exceeded and the UFSAR analysis of the event remained bounding.

The Start Bus PT fuse that failed was a Bussman Limitron KTN-6. Investigation determined that the fuse failed due to mechanical separation of end plate from end cap due to inadequate manufacturing aggravated by age. An extent of condition evaluation determined that similar components include Bussman KTN-6 as well as KTN (R) and KWN (R) fuses with a rating of less than 30 amps. A circuit evaluation of installed fuses of this type determined that a failure of these fuses would not inhibit or prevent the actuation of a safety function. Hence the installed fuses could not create a substantial safety hazard and this event is not reportable under 10 CFR Part 21.

## V. ASSESSMENT OF SAFETY CONSEQUENCES

Based on the above "Analysis of the Event," this event did not adversely affect the health and safety of plant personnel or the general public.

| 1. FACILITY NAME | 2. DOCKET | 6. LER NUMBER | | | 3. PAGE |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REV NO. | |
| Sequoyah Nuclear Plant Unit 1 | 05000327 | 2011 | -- 005 | -- 01 | 6 OF 6 |

NARRATIVE

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions:

Control room personnel responded to the reactor trip as prescribed by emergency procedures. The failed Start Bus potential transformer fuse was replaced and the 1A Start Bus was re-aligned to normal supply.

B. Corrective Actions to Prevent Recurrence:

Corrective actions to prevent recurrence will include creating preventative maintenance procedures to address replacing fuses in locations that are trip risk sensitive or could result in engineered safety feature actuations.

VII. ADDITIONAL INFORMATION

A. Failed Components:

The failed component was a Bussman Limitron KTN-6 fuse.

B. Previous LERs on Similar Events:

A review of previous reportable events for the past three years did not identify any previous similar events.

C. Additional Information:

The corrective action document for this report is Problem Evaluation Report 419705.

D. Safety System Functional Failure:

This event did not result in a safety system functional failure in accordance with 10 CFR 50.73(a)(2)(v).

E. Unplanned Scram with Complications:

This event did not result in an unplanned scram with complications.

VIII. COMMITMENTS

None.